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OCT 1 2 2004

LR-N04-0426

PSEG
Nuclear LLC

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING REQUEST FOR AUTHORIZATION TO USE A RISK-INFORMED INSERVICE INSPECTION ALTERNATIVE TO THE ASME BOILER AND PRESSURE VESSEL CODE SECTION XI REQUIREMENTS FOR CLASS 1 AND 2 PIPING
HOPE CREEK GENERATING STATION
DOCKET NO. 50-354**

Reference: LR-N04-0366, Response To Request For Additional Information Regarding Request For Authorization To Use A Risk-Informed Inservice Inspection Alternative To The ASME Boiler And Pressure Vessel Code Section XI Requirements For Class 1 And 2 Piping, dated August 17, 2004

NRC letter dated August 5, 2004, requested additional information regarding a proposed Alternative to utilize a Risk-Informed Inservice Inspection Plan for the Hope Creek Generating Station. PSEG provided the requested information in the referenced letter. These letters resulted in the NRC staff verbally requesting additional clarification. Attachment 1 contains PSEG's revised responses to questions 1, 6, and 7 of the August 5, 2004 NRC RAI.

If you have any questions or require additional information, please contact Mr. Michael Mosier at (856) 339-5434.

Sincerely,

John Carlin
Vice President – Nuclear Assurance

Attachment

A047

95-2168 REV. 7/99

OCT 1 2 2004

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HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. NFP-57
DOCKET NO. 50-354
REQUEST FOR ADDITIONAL INFORMATION

NRC Question 1:

Regulatory Guide (RG) 1.178, *An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping, Revision 1, dated September 2003, replaced the original "For Trial Use"* RG dated September 1998. Revision 1 of RG 1.178 includes guidance on what should be included in risk-informed inservice inspection (RI-ISI) submittals, particularly in dealing with probabilistic risk assessment (PRA) issues. Specifically, on page 28 of RG 1.178, the following is stated:

> "A description of the staff and industry reviews performed on the PRA. Limitations, weakness or improvements identified by the reviewers that could change the results of the PRA should be discussed. The resolution of the reviewer comments, or an explanation of the insensitivity of the analysis used to support the submittal to the comment, should be provided."

Section 1.2 of your submittal discussed the HCGS IPE. By letter dated April 23, 1996, the NRC issued a safety evaluation, concluding that the IPE had met the intent of GL88-20, and had identified plant-specific vulnerabilities per the guidance of NUREG-1335. With regard to the IPE, answers to the following are required:

a. What weaknesses were identified?

b. What was done to correct the identified weaknesses or why the uncorrected weaknesses are not relevant to this application?

PSEG Response to Question 1 (Revised):

As discussed in the referenced letter PSEG, addressed how a plant design vulnerability identified by the IPE related to long-term loss of Heating, Ventilation and Air Conditioning (HVAC) in a limited number of electrical equipment areas. To address this vulnerability, a procedure (HC.OP-AB.HVAC-0001) was implemented in 1996 instructing station personnel to provide alternative cooling methods such as opening doors and bringing in portable fans when the normal cooling mode is not available.

In the NRC's Safety Evaluation, it was pointed out that the loss of switchgear or panel room HVAC had a core damage frequency (CDF) of about 3.3E-03/year before the procedure was implemented. After crediting the procedure in the PRA, this CDF changed to about 1.0E-06/year. The Safety Evaluation stated that, "the staff believes the reduction of this sequence by three orders of magnitude may be overly optimistic." While the Safety Evaluation concluded that this issue did not have an effect on the overall validity of the IPE, it was recommended that the related analysis be re-examined to verify these calculated CDFs.

For the purpose of this submittal, PSEG addressed why the analysis used to support this submittal is insensitive to the weakness.

The Consequence Evaluation performed in support of this submittal follows the guidance of EPRI TR-112657 Rev. B-A. Piping segments are evaluated in three ways:

- Per Section 3.3.3.1 in the TR, pipe breaks that cause Initiating Events modeled in our PRA are quantitatively evaluated. Refer to Table 3-4.
- Other pipe segments such as in standby systems are evaluated using "look-up" tables in the TR.
- Some pipe segments are evaluated by both methods as discussed in Section 3.3.3.3 and Table 3-13 of the TR.

For those pipe segments that were quantitatively evaluated using our Hope Creek PRA Rev. 1.3, a sensitivity study has been performed to demonstrate that the weakness discussed above is not significant to the application. For pipe segments that were evaluated using "look-up" tables, our PRA was not quantitatively used to calculate conditional core damage probability (CCDP) and loss of ventilation is an insignificant contributor to credited backup trains in the evaluation (the sensitivity study described below further supports this judgment).

The Sensitivity Study:

There are four Human Error Probabilities (HEPs) used in the Hope Creek PRA Rev. 1.3 related to the procedure put in place to mitigate loss of HVAC in the Switchgear and Panel Rooms. They are:

PRA basic event	Description	Probability
NR-HVAC-12	FAILURE TO RECOVER IN 12 HOURS	1.0E-3
NR-HVC-PNRM-12	FAILURE TO PROVIDE ALTERNATE VENTILATION TO 1E PANEL ROOM IN 12HOURS	3.0E-4
NRHVCPNRM12-01	SCREEN VALUE FOR NR-HVC-PNRM-12	1.0E-1
NRHVCSWGR24-01	SCREEN VALUE FOR FAILURE TO RESTORE SWGR ROOM COOLING	1.0E-1

The first basic event is used for a loss of HVAC initiator, and the others are used for loss of HVAC while the plant is being shut down due to another initiator. The value of the second basic event in the above table was derived using the EPRI methodology for Human Reliability Analysis and appears to be overly optimistic as discussed in the NRC's Safety Evaluation. The value of the last two basic events are "screening values" that appear to be overly conservative given the implementation of the procedure. The value of the remaining basic event (first line item in the table above) is not well documented, but based on recent HRA work it is somewhat more realistic than the second basic event. However, it is still judged to be somewhat optimistic.

To evaluate the impact of these HEPs on this application, the first and second basic events were multiplied by a factor of ten and then the model was re-quantified. The last two basic events were not changed for this sensitivity study. The before and after results are shown in Table 1 below, in a format similar to TR Table 3-4.

Conclusion:

The sensitivity study demonstrates that the HEPs associated with the procedure, implemented to address the IPE identified vulnerability, do not have significance to this application.

Table 1

Base Case

Initiating Event	Frequency [1/rx-yr]	CDF [1/rx-yr]	CCDP	LERF [1/rx-yr]	CLERP	LERP	Consequence
A - Large LOCA	3.0E-05	3.9E-09	1.3E-04	1.2E-10	4.0E-06	3.0E-02	High
S1 - Medium LOCA	9.3E-04	1.7E-07	1.9E-04	3.8E-09	4.0E-06	2.0E-02	High
S2 - Small LOCA	6.5E-04	5.6E-08	8.6E-05	1.6E-10	2.5E-07	<1.0E-02	Medium
TB - Break Outside Containment	2.1E-02	1.4E-08	6.9E-07	7.4E-11	3.5E-09	1.0E-02	Low
TC - Loss of Condenser	2.0E-01	2.3E-06	1.1E-05	2.4E-07	1.2E-06	1.0E-01	Medium
TF - Loss of Feedwater	8.5E-02	8.7E-07	1.0E-05	1.0E-07	1.2E-06	1.1E-01	Medium
TSD - Manual Shutdown	5.0E-01	1.8E-08	3.7E-08	5.8E-11	1.2E-10	<1.0E-02	Low
TT - Turbine Trip	1.4E+00	1.2E-06	8.6E-07	1.3E-07	8.9E-08	1.0E-01	Low

NR-HVAC-12 changed to 1.0E-02 and NR-HVC-PNRM-12 changed to 3.0E-03

Initiating Event	Frequency [1/rx-yr]	CDF [1/rx-yr]	CCDP	LERF [1/rx-yr]	CLERP	LERP	Consequence
A - Large LOCA	3.0E-05	3.9E-09	1.3E-04	1.2E-10	3.9E-06	3.0E-02	High
S1 - Medium LOCA	9.3E-04	1.7E-07	1.9E-04	3.8E-09	4.0E-06	2.0E-02	High
S2 - Small LOCA	6.5E-04	5.6E-08	8.6E-05	1.6E-10	2.5E-07	<1.0E-02	Medium
TB - Break Outside Containment	2.1E-02	1.4E-08	6.9E-07	7.4E-11	3.5E-09	1.0E-02	Low
TC - Loss of Condenser	2.0E-01	2.3E-06	1.1E-05	2.4E-07	1.2E-06	1.0E-01	Medium
TF - Loss of Feedwater	8.5E-02	8.7E-07	1.0E-05	1.0E-07	1.2E-06	1.1E-01	Medium
TSD - Manual Shutdown	5.0E-01	1.9E-08	3.8E-08	5.8E-11	1.2E-10	<1.0E-02	Low
TT - Turbine Trip	1.4E+00	1.2E-06	8.6E-07	1.3E-07	8.9E-09	1.1E-01	Low

Table 1 (cont'd)

The only changes are:

- TSD – Manual Shutdown CDF changed from 1.8E-08/year to 1.9E-08/year. Insignificant change. The CDF is used to derive the CCDP. See the next item.
- TSD – Manual Shutdown CCDP changed from 3.7E-08 to 3.8E-08. Insignificant change. This does not challenge the "Low" to "Medium" threshold value for CCDP = 1.0E-06. (TR Table 3-1)
- A – Large LOCA CLERP changed from 4.0E-06 to 3.9E-06 due to rounding off. No change.
- TT – Turbine Trip LERP is changed from 1.0E-01 to 1.1E-01. The base value was actually rounded down to 1.0E-01 from 1.1E-01. No change.

NRC Question 6:

Section 3 of PSEG's March 1, 2004 submittal states that the RI-ISI program for HCGS will deviate from the EPRI RI-ISI methodology for the assessment for thermal stratification, cycling, and striping (TASCS). State whether or not the revised methodology for assessing TASCS potential is in conformance with the updated criteria described in the EPRI letter to the NRC dated March 28, 2001. Also, confirm that as stated in the March 28, 2001 letter (Available under ADAMS Accession Number ML011070238), once the final material reliability program guidance has been developed, the RI-ISI program will be updated for the evaluation susceptibility to TASCS, as appropriate.

PSEG Response to Question 6 (Revised):

The last paragraph of Section 3.0 states, in part, "The above criteria have previously been submitted by EPRI for generic approval (Letters dated February 28, 2001 and March 28, 2001, from P.J. O'Regan (EPRI) to DR. B. Sheron (USNRC), "Extension of Risk-Informed Inservice Inspection Methodology"). The methodology used in the Hope Creek RI-ISI application for assessing TASCS potential conforms to these updated criteria."

Once the final Materials Reliability Program (MRP) guidance has been developed, PSEG will review and update the Hope Creek RI-ISI program as appropriate for assessing the TASCS potential. The program will be in accordance with the latest NRC approved guidance.

NRC Question 7:

Section 2.2 of the submittal states, in part, "the feedwater nozzle-to-safe end weld locations are included in the scope of both the NUREG 0619 Program and the RI-ISI Program. The plant augmented inspection program requirements for these locations are not affected or changed by the RI-ISI Program." Explain if credit has been taken from this augmented program as part of the RI-ISI program. If so, explain the weld selection criteria as compared to EPRI TR-112657, given that NUREG-0619 is not considered as an augmented program in EPRI TR-112657.

PSEG Response to Question 7 (Revised):

Section 2.2 of the template submittal documents all existing plant augmented inspection programs that address common piping with the RI-ISI application. As stated above, the feedwater nozzle-to-safe end weld locations are included in the scope of both the NUREG-0619 Program and the RI-ISI Program. The plant augmented inspection program requirements for these locations are not affected or changed by the RI-ISI Program. The damage mechanism identified for these locations is crevice corrosion due to the presence of a thermal sleeve, oxygen and high temperatures (per EPRI TR-112657). These locations will be subjected to an appropriate examination per EPRI TR-112657 in addition to the examinations performed per the plant's NUREG-0619 Program. The industry is currently in the process of clarifying the definition of the crevice corrosion degradation mechanism. Once the clarification is issued, Hope Creek will evaluate the impact of the new information on the RI-ISI program.